NON-PUBLIC?: N
ACCESSION #: 9002220365
LICENSEE EVENT REPORT (LER)

FACILITY NAME: NORTH ANNA POWER STATION UNIT 1 PAGE: 1 OF 6

DOCKET NUMBER: 05000338

TITLE: REACTOR TRIP ON STEAM/FEEDWATER FLOW MISMATCH DUE TO A FAILED
DRIVER CARD ON A FEEDWATER REGULATING VALVE
EVENT DATE: 10/23/90 LER #: 90-001-00 REPORT DATE: 02/08/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: G. E. KANE, STATION MANAGER TELEPHONE: (703) 894-5151

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: FCV MANUFACTURER: C635
X SB TCV C635
X IG DET W120
X SB ISV C635
X EL BKR B455
X SJ RV S102
X SJ RV C710
REPORTABLE NPRDS: Y
Y
Y
Y
Y
N
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1522 hours on January 23, 1990, Unit 1 experienced an automatic
trip from 100 percent power. The initiating signal for the

reactor trip was a low level in the "C" Steam Generator with a steam flow greater than feedwater flow mismatch. The mismatch resulted from closure of the 'C' Main Feedwater Regulating Valve. The closure was caused by a failed printed circuit driver card in the valve controller. After event investigation and corrective action, Unit 1 was returned to critical on January 24, 1990 at 0241 hours.

This event constitutes an automatic actuation of the Reactor Protection System and is reportable pursuant to 10 CFR 50.73 (a)(2)(iv).

No significant safety consequences resulted from the reactor trip because plant safety systems functioned as designed. The Reactor Coolant System parameters stabilized at their normal post trip values. There was no release of radioactive materials due to the trip.

The health and safety of the public were not affected at any time during this event.

END OF ABSTRACT

1.0 Description of the Event

At 1522 hours on January 23, 1990, Unit 1 experienced an automatic trip from 100 percent power. The initiating signal for the reactor trip was a low level in the "C" Steam Generator (S/G) (EIIS System Identifier AB, Component Identifier SG) with a steam flow greater than feedwater flow mismatch. The mismatch resulted from closure of the 'C' Main Feedwater Regulating Valve . The closure was caused by a failed Westinghouse 7300 printed circuit driver card in the valve's electronic control system. This event constitutes an automatic actuation of the Reactor Protection System and is reportable pursuant to 10 CFR 50.73 (a)(2)(iv). A four hour report was made to the NRC at 1725 hours on January 23, 1990, in accordance with 10 CFR 50.72 (b)(2)(ii).

Control Room Operators responded to the reactor trip in accordance with Emergency Operating Procedure EP-0, "Reactor Trip or Safety Injection". The plant reponded as expected with Pressurizer pressure decreasing to 1950 Psig/

pressurizer level decreasing to 21% and Reactor Coolant System (RCS) temperature decreasing to 543.5 degrees F. Following evaluation of the RCS parameters and indications, Control Room personnel transitioned from EP-0 to ES-0.1 "Reactor Trip Response".

Subsequent to the reactor trip, the "C" phase of the electrical generator output breaker (G-12) control logic indicated closed due to a dirty contact, although the breaker had actually opened. As a result, the switchyard breakers G102 and G1T5H opened as designed to isolate the turbine generator from the electrical grid. Station service electrical buses automatically transferred to the reserve station service transformers. At 1525 hours, Control Room Operators entered Abnormal procedure AP-10.1, "Loss of Electrical Power" to identify the electrical malfunction and verify electrical system lineups.

While monitoring intermediate range neutron flux, Control Room Operators observed that intermediate range detector N-35 was undercompensated, preventing automatic reinstatement of the source range detectors. At 1538 hours, the operators manually reinstated the source range detectors in accordance with ES-0.1.

1.0 Description of the Event cont'd.

Plant Equipment responded as expected with the following exceptions:

o The 'C' Main Feedwater Regulating Valve, 1-FW-FCV-1498 (EIIS System Identifier SJ, Component Identifier FCV, Vendor Identifier C635, Model Number D100-12), failed closed due to a failed printed circuit driver card.

o The Intermediate Range Nuclear Instrument (N-35) (EIIS System Identifier IG, Component Identifier DET, Vendor Identifier W120) was undercompensated.

o The Condenser Steam Dump 'B' (1-MS-TCV-1408B) (EIIS System Identifier SB, Component Identifier TCV Vendor Identifier C635 Model Number 8-RA36RG) indicated mid position after closure.

o The 1 Reheat Right Stop Valve (EIIS System Identifier SB, Component Identifier ISV, Vendor Identifier C635, Model Number D100-160-3) had a broken indicator arm.

o The 'C' Phase Generator Output Breaker (G-12) (EIIS System Identifier 41, Component Identifier BKR, Vendor Identifier B455) failed to indicate open.

o The tube side relief valves on the 2A and 4B feedwater heaters (1-RV-SV-112A and 114B) (EIIS System Identifier SJ, Component Identifier RV, Vendor Identifier C710) opened.

o The suction relief valve (1-FW-RV-102C) on the 'C' Main Feedwater Pump (EIIS System Identifier SJ, Component Identifier RV, Vendor Identifier S012, Model Number 451132-B) opened.

After event investigation and corrective action, Unit 1 was returned to critical on January 24, 1990 at 0241 hours.

2.0 Significant Safety Consequences and Implications

No significant safety consequences resulted from the reactor trip because plant safety systems functioned as designed. The Reactor Coolant System parameters stabilized at their normal post trip values. There was no release of radioactive materials due to the trip.

2.0 Significant Safety Consequences and Implications Cont'd

The health and safety of the public were not affected at any time during this event.

3.0
Cause of the Event

The cause of this event was failure of the printed circuit driver card on the 'C' Main Feedwater Regulating valve which caused valve closure. Preliminary investigation has indicated that the driver card failed due to age. VEPCO is conducting an engineering evaluation and root cause investigation to discern any contributing factors to the failure and determine if other driver cards of similar vintage have an increased failure risk.

Upon completion of the engineering and root cause evaluation, necessary corrective actions will be implemented.

4.0 Immediate Corrective Action

As an immediate corrective action, Emergency Operating Procedure EP-0, "Reactor Trip or Safety Injection", was entered and the plant stabilized in Hot Standby.

5.0 Additional Corrective Actions

The following corrective actions were taken to correct the hardware problems that occurred during this event:

o The 'C' Main Feedwater Regulating Valve, 1-FW-FCV-1498 failed driver card was replaced. A functional test was performed on the similar driver cards for the 'A' and 'B' Main Feedwater Regulating Valves. The cards and associated drivers were verified to be operating properly.

o The Intermediate Range Nuclear Instrument (N-35) compensation voltage was adjusted in accordance with Westinghouse methodology.

o The Condenser Steam Dump 'B' (1-MS-TCV-1408B) limit switch was adjusted.

o The Broken indicator arm on 1 Reheat Right Stop Valve was repaired.

o The Generator output Breaker (G-12) phase to "C" indication contact was cleaned. As a preventive measure, the Generator Output Breaker phase "A" and "B" indication contacts were also cleaned.

o The Tube side relief valves on the 2A and 4B feedwater heaters (1-RV-SV-112A and 114B) reseated.

o The Suction relief valve (1-FW-RV-102C) on 'C' Main Feedwater Pump reseated.

6.0 Actions to Prevent Recurrence

Engineering evaluations will be conducted on the the following malfunctions:

- o Intermediate Range Monitor N-35 undercompensation
- o Failed driver card on 'C' Main Feedwater Regulating Valve, 1-FW-FCV-1498.
- o Lifting of the tube side relief valves on the 2A and 4B feedwater heaters (1-RV-SV-112A and 114B).
- o Lifting of the suction relief valve (1-FW-RV-102C) on the 'C' Main Feedwater Pump.

Corrective actions will be taken as required as a result of these evaluations.

7.0 Similar Events

Previous reactor trips due to steam flow greater than feedwater flow mismatch coincident with a low steam generator level occurred on Unit 1: May 20, 1986 (LER-N1-86-008), August 6, 1988 (LER N1-88-020), February 25, 1989 (LER N1-89-005) and on Unit 2: March 13, 1984 (LER N2-84-001). June 25, 1984, (LER N2-84-005) and June 29, 1986 (LER N2-88-009)

The Reactor Trip reported in LER N1-89-005 was also caused by the failure of the 'C' Main Feedwater Regulating Valve. However, the cause of the regulating valve failure for that event was a broken air line to the valve.

Licensee Event Report Failure Continuation appended on LER Form.

ATTACHMENT 1 TO 9002220365 PAGE 1 OF 1

10 CFR 50.73
VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION

P. O. BOX 402

MINERAL, VIRGINIA 23117

February 8, 1990

U. S. Nuclear Regulatory Commission Serial No. N-90-001
Attention: Document Control Desk NAPS/CSW:csw
Washington, D.C. 20555 Docket No. 50-338

License No. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. LER 90-01-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very Truly Yours,

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Mariena Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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